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**SOME CHARACTERISTICS OF RADIOISOTOPE
POWER SOURCES IN AN OCEAN ENVIRONMENT**

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Santa Monica, California

March 1974

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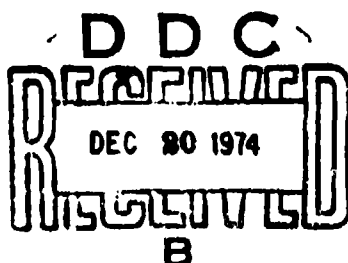
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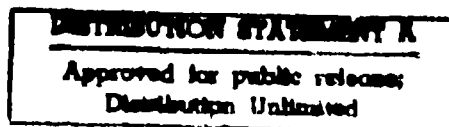
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Abstract

The results of this study show that lightweight radioisotope heat sources can be considered for use undersea with little shielding necessary for electronics packages in close proximity to the heat source. Acceptable integrated electronic dose levels of 10^7 rads for gamma irradiation and 10^4 reps for neutron bombardment were obtained for extended mission times for the three isotopes considered. It is unnecessary to shield the isotope ^{210}Po for electronic protection. However, use would be restricted because of its short half-life and this isotope could present special heat management problems not encountered with either ^{60}Co or ^{244}Cm . For ^{244}Cm and ^{60}Co it was found that the shield around the electronics package becomes important due to the strong back scattering effect of neutrons and gammas from the surrounding seawater. The ^{244}Cm heat source appears to require less shielding to electronics level and would therefore take up less volume in the undersea system than the ^{60}Co heat source. Recovery of an isotope-powered undersea electronics package must be done remotely. It will be necessary to provide shielding to protect personnel engaged in the retrieval of the package (and also in the initial launching) and this vehicle cask will also have to provide for auxiliary cooling of the radioisotope heating source.

I. INTRODUCTION

For remote operation of submerged sensor packages, auxiliary power can be provided in the kilowatt range by radioisotope heat sources. This power can be used to either operate electronics sensor packages or to operate small power plants used for propulsion and stabilization. The goals of a useful mission life, reusability, mechanical simplicity, and reliability are consistent with this choice of radioisotope heat source. The problems of handling and safety must, however, be considered in any application of these materials.

In this paper, an analysis of the shield requirements of several isotopes is presented. Also the relative biological hazard of the isotope is discussed, should a release occur. The purpose of this study is to establish the minimum size (in terms of weight and volume) of a shielded radioisotope heat source that can possibly be considered for use in an undersea environment. This size restriction is necessary for structural reasons in deep ocean locations and also in mobile systems to reduce vehicle volume requirements. The thermal energy produced by the source can be utilized in any number of converters (thermoelectric, Brayton, Stirling, or organic Rankine) to generate electrical or shaft power for various uses. Since we were looking for the most compact heat source currently conceivable, examination of radioisotopes available indicated three primary candidates -- ^{60}Co , ^{244}Cm and ^{210}Po . Consideration of many other isotopes led to their elimination because of failure to satisfy one of the requirements cited above. If future variations in mission goals,

isotope availability or cost occur, then others such as ^{170}Tm , ^{90}Sr , or ^{238}Pu may be reconsidered.

Hence we are examining the possible isotopes that could be used to provide a very large power requirement undersea, where shielding via distance is unacceptable. For any detailed design, one would utilize more sophisticated shield/dose codes to quantify shield material and geometry. It is our intention to show that many of the isotopes can be eliminated based on relatively simple calculations at the preliminary design stage. The uniqueness of this discussion lies in its focus on remote undersea systems having millions of curves on board where shielding via distance is unacceptable.

To have a common basis on which to study the characteristics of each radioisotope, a source strength of 30-35 kw was assumed, as well as a mission duration of 2000 hours and shielding necessary to electronics levels only. Our concern then was to determine the minimum source weight and size including necessary shielding for a useful electronics payload situated in close proximity to the heat source to minimize the overall size and weight of the entire system.

II. DISCUSSION

A. Cobalt 60

^{60}Co is a strong γ emitter which requires heavy shielding for use near personnel and electronics. However, because of its long half-life (5.24 years), high availability ($\sim 10 - 18 \times 10^6$ curies of 300-400 curies/gm ^{60}Co currently available from the A.E.C. at Savannah River), low cost ($\sim \$15/\text{watt}$) and high specific activity (20-60 watts/cm³), ^{60}Co becomes

a prime candidate for consideration in underwater power systems. The major problem associated with the use of ^{60}Co , however, is the requirement for extensive shielding. To examine this problem, a shielding analysis for a ^{60}Co source was performed at Los Alamos Laboratories using a two-dimensional transport theory code. Preliminary one-dimensional calculations indicated that it might be possible to shield a 30 kw_t ^{60}Co heat source to approximately electronics levels ($\sim 10^7$ Rads integrated dose) within the weight limitations of small systems. However, these calculations could not account for the back scatter of γ radiation from the surrounding seawater so a 2-dimensional study became necessary.

The geometry used in this study is shown in Figure 1. The heat source is cylindrical and is surrounded by a tungsten shell (~ 1.75 cm thick) to capture $\sim 90\%$ of the γ radiation emitted from the source. A cylindrical shadow shield of ^{238}U is placed at the front of the source and the electronics payload region is shielded along its front face and partially along the sides by ^{238}U . This barrier is used to shield against γ radiation backscatter from the water. The LiH shield shown in Figure 1 would be of use in neutron absorbing situations and the iron in the cylinder is meant to approximate a powerplant in the actual system. The LiH and Fe shields were omitted from these calculations ($t_1 = t_2 = 0$). The ^{60}Co source was taken to be 30 kw_t with a power density of 40 w/cm^3 . The weight of the ^{60}Co and tungsten shell for this geometry is then ~ 65 lb. If the power density is taken to be 60 w/cm^3 (more costly to produce and less available) and the tungsten is reduced to a 1.5 cm thickness, the

source weight becomes ~40 lb. In any case, it is clear that the ^{60}Co source with the tungsten shell will weigh between 40 and 60 lb. allowing for no internal coolant volume. Therefore, this is a lower limit to source weight.

The results of the two-dimensional transport calculations (using P_0 scattering, S-8 quadrature and 3 energy groups) indicated that t_3 could be varied from 1.0 to 5.0 cm and R_3 from 7.0 to 19.0 cm with little effect on radiation levels in the payload region. Backscattering of the γ 's from the water was thus found to have a major effect on the γ radiation levels in the electronics payload region. The separation distance from the source to the electronics payload (the distance z shown in the diagram) and the payload shield thickness are the dominant factors in determining the radiation levels in the payload region.

The payload volume shielded to a level of 10^7 Rads integrated dose over a 2,000 hour mission life is shown in Figure 2 as a function of payload shield weight. This is for a 3.0 ft. separation distance. The shadow shield is 2.5 cm thick in this case and weighs ~15.0 lb. The variation in mission time as a function of payload shield weight for a 3 foot separation distance and integrated dose of 10^7 Rads is shown in Figure 3. If the separation distance is increased, in some way, to 5.0 feet the results are as shown in Figure 4. One can see that there are attractive combinations of shield volume and shield weight for mission times of the order of several thousand hours.

This shielding analysis indicates the extreme limitations placed on allowed mission time or shielded electronics volume for a ^{60}Co radioisotope

system. In addition there are other factors to be considered. Even though it may be possible to shield a small volume for a short time to dose levels that might be tolerable for electronics, these dose levels are orders of magnitude larger than that which can be tolerated by man. For a 30kw_t source, it takes approximately 6.5 to 7.0 in. of tungsten to shield to human tolerance (~ 30 mr/hr). Retrieval times of several hours would be expected, relaxing this tungsten requirement somewhat. This would permit larger short time doses that could be averaged over calendar time to allow for convenient recovery and maintenance. It would probably be necessary to shield the complete system during recovery as well as to provide auxiliary cooling for the heat source. One would expect these requirements to be met by using the technology developed in fuel element transport casks. Environmental hazards are also great if by chance the ^{60}Co source were to come into contact with people. The ^{60}Co dose rate in air for a 30kw_t source (90% shielded) is shown in Figure 5. Note that at 100ft. it takes 3.0 hrs. to reach a dose level of 600 Rads.

B. Polonium 210

The radioisotope ^{210}Po is an α emitter, and the secondary radiation associated with its decay is of low intensity (a soft γ of .8 Mev). Thus, it does not have to be shielded for use near electronics. However, when large quantities of ^{210}Po are assembled a small amount of shielding must be provided for the protection of personnel using this heat source. ^{210}Po was considered as a fuel for the SNAP-29 and Poodle programs and its fuel forms have been extensively studied by Mound Laboratories. The matrix fuel form developed by Mound has a very high power density (40 - 80 w/cc)

and has been operated extensively at temperatures between 2000°F and 3000°F.

Several problems are associated with the use of ^{210}Po . Its availability is difficult to estimate. Current production levels at Savannah River and Mound Laboratory are very low, averaging several grams (100's of watts) per year. At present, there are no plans to increase this production capability.

Even though the short half-life of ^{210}Po is a disadvantage when considering thermal management within the heat source, initial inventory costs, and system scheduling, it is an advantage in terms of possible environmental effects. Its rapid decay means that if, as a result of an accident, substantial amounts of the ^{210}Po escape into the environment it will take only a relatively short time for the effect of the accident to die away. In addition, there are encapsulation techniques which can be used to insure that negligible amounts of ^{210}Po are released to the environment over a period of, for example, 10 half-lives even if a catastrophic failure occurs. As an example, Battelle-Northwest has proposed encapsulating ^{210}Po in a microsphere fuel form which has an outside layer that can be made very resistant to seawater corrosion. For this fuel form, a fuel element failure without melting would result only in a release of microspheres which in themselves are coated with oxidation resistant clads. Release of ^{210}Po into the environment would require the failure of thousands of small capsules. Thus, a safety analysis and a study of possible failure modes for the heat source become of prime importance for this radioisotope heat source. The major drawback of polonium is its 138 day half-life. This fact alone gives one a difficult but

not insurmountable heat management problem for missions longer than several months.

C. Curium 244

Using the transport theory code TWOTRAN², multigroup calculations were performed on ²⁴⁴Cm to determine the neutron and gamma shield requirements for several relevant geometries and power levels. (²⁴⁴Cm emits both neutrons and gamma rays, and can also be formed into a critical configuration.) These calculations were carried out separately for the neutron attenuation and the gamma attenuation problems. No secondary gamma production was considered. Other assumptions were made and these will be discussed as appropriate. Curium would be incorporated as the oxide, Cm₂O₃, within the heat source. Several properties for this compound are listed in Table I.

Radiological properties for ²⁴⁴Cm are listed in Table II. The gamma rate in Table II is a summary of that given in ORNL 4357 (1) where a more refined group structure is listed. The predominant gamma arises from the decay of ²⁴⁴Cm, with lower intensities due to spontaneous fission, fission product decay, and impurity decay.

Spontaneous fission accounts for the major portion of the neutrons and the peak energy of these neutrons is taken to be 0.6 MeV as shown in Figure 6. Alpha particle reactions with oxygen contribute approximately 3.5% of the neutrons at a peak of about 2.5 MeV (see Figure 6). Hence the approximate mean neutron energy of 1.5 MeV is listed in Table II and this level was used for the calculation.

The curium heat source is approximated by a cylinder having a radius of 4 inches and a length of 8 inches, and self shielding within the heat

source is neglected. The environment surrounding the source and its container is water. A simple hand calculation indicates that water reflection of source neutrons should provide a large component in the electronics region.

Significant to the use of ^{244}Cm is the impact an uncontrolled release would have on the environment. Cm_2O_3 appears to be insoluble in water. Hence during an accident the oxide pellets would either settle to the ocean bottom, or small particles would remain in suspension. Its half life of 18.1 years means that it will take hundreds of years to decay away. When it does decay it becomes ^{240}Pu which has a half life of 6,760 years and which can lead to a serious biological impact. ^{210}Po on the other hand decays to ^{206}Pb , which is a non-radioactive isotope.

Shield Geometry and Materials

The limiting dose to electronics from a curium heat source is expected to be that due to neutrons. To remove neutrons one must first moderate and second capture them. Lithium hydride was the material considered for the task of moderating the source neutrons. A cadmium sheet was placed around the electronics payload to absorb thermal neutrons reflected from the surrounding ocean. The geometry of the system is shown in Figure 7.

In the numerical calculations, the source/payload separation distance, LiH thickness and shape, and the ^{238}U thickness (for gamma attenuation) were varied.

Limiting Electronic Dose Levels

The limiting electronic dose level is 10^7 rads for gamma radiation. This is equivalent to a neutron dose of 10^4 reps. Where a component sees both types of radiation the separate effects should be appropriately summed.

To convert gamma radiation fluence levels into rads the following equivalence was assumed:

$$10^7 \text{ rads} \Leftrightarrow 10^{15} \gamma/\text{cm}^2$$

Three energy groups were assumed for the gamma calculations. They were

Group	Energy Interval MeV	Source $\gamma/\text{watt sec}$
1	0.5 - 1.0	.267(10 ⁸)
2	0.3 - 0.5	1.5 (10 ⁸)
3	0.0 - 0.3	1.5 (10 ⁸)

Appropriate gamma source strengths in each of these energy groups are also listed.

The neutrons were expected to thermalize and this necessitated the following group structure. Also shown is the neutron source strength (which is non-zero only in the fast group).

Group	Energy Interval MeV	Source n/watt sec
1	10 ev - 10 MeV	4.07(10 ⁶)
2	1 ev - 10 ev	0
3	0.5 ev - 1.0 ev	0
4	0.001 ev - 0.5 ev	0

To convert fluence data into dose levels it was assumed that for the latter three groups,

$$300 \frac{\text{n}}{\text{cm}^2 \text{ sec}} \Leftrightarrow \frac{1 \text{ mr}}{\text{hr}}$$

This led to

$$1.08(10^9) \frac{n}{cm^2} \Leftrightarrow 1 \text{ rep (grps 2,3,4)}$$

In group 1 a mean energy, assuming an E^{-1} slowing down spectrum, of 0.1 MeV yielded

$$\frac{.53n}{cm^2 \text{ sec}} \Leftrightarrow \frac{1 \text{ mr}}{hr}$$

This becomes

$$1.91(10^8) \frac{n}{cm^2} \Leftrightarrow 1 \text{ rep (grp 1)}$$

The above numerical equivalences were taken from Rockwell (2) and ORNL 3576 (3).

Numerical Results

During the calculations with TWOTRAN, the source and electronics payload region were kept constant in size. The cylindrical shell of water began at 19.5 cm radius and extended radially outward. Water was assumed to extend back from the source and also back from the 1/16 in. cadmium sheet behind the payload, even when the source/payload separation distance was varied. The calculational data are summarized in Table III for neutron attenuation and in Table IV for gamma attenuation.

The data from Table IV are sketched in Figure 8. By inspection it is seen that the induced dose is below damage level by a factor of 10 even for the 2 ft separation case. Hence no uranium is needed to shield

electronics from the curium gamma source.

From the neutron attenuation results given in Table III, it is clear that with a simple shield in front of the source, the Cd sheet reduces the dose in the payload region by a factor of 2. The effect of shield thickness on neutron attenuation is given in Figure 9. From this figure it is clear that most of the dose is due to neutrons which scatter back from the water, having thus passed around the source face shield. The lowest curve is obtained by placing a one inch LiH shield around the source cylinder. It seems possible then to reduce the neutron dose level to approximately $1.5(10^4)$ reps with a one inch LiH shield at the front and around the side of the source cylinder.

Shield Weight

Since no ^{238}U shield is needed, the shield weight is made up of a 1 inch shell of LiH around the source and a 1/16 inch shell of Cd around the electronics payload. The shield weights are as follows:

Cd	13 lbs
LiH	8 lbs

Thus the minimum shield weight for a ^{244}Cm source will lie somewhere between 20 and 25 lbs.

Since ^{244}Cm is an intense neutron emitter, the shielding necessary for handling by humans is as great as for ^{60}Co in terms of weight. Availability is also a problem. All of the ^{244}Cm produced so far has been made at Savannah River by irradiation, and then separated and formed into fuel elements at Oak Ridge. The current cost for this process is approximately \$1000/watt.

To obtain ^{244}Cm at a lower cost, it must be recovered from spent and reprocessed reactor fuel elements. Estimates indicate that it will be 1978 to 1985 before large amounts of ^{244}Cm will be available by this method. Initial recovery could be performed at Oak Ridge, and this might yield 10-20 kg $^{244}\text{Cm}/\text{year}$ (14 kg are needed for one 32 kw heat source) if the present facilities were to be upgraded. The estimated production cost for recovering ^{244}Cm from spent reactor fuel is approximately \$100/watt encapsulated.

III. Conclusions

Use of ^{210}Po would be difficult for missions of several thousand hours duration because its 138 day half-life leads both to severe heat management and logistics problems. Extreme limitations on electronics payload size and/or mission time exist for small systems using a ^{60}Co heat source where shielding via distance is unacceptable. ^{244}Cm can provide the energy required for small undersea systems if neutron shields are used and if neutron return from the ocean is counteracted by electronics region shielding. ^{244}Cm can be made available from power reactor fuel reprocessing. Current plans do not call for such recovery, but steps can be taken to produce ^{244}Cm in the reprocessing cycle when the need exists.

Acknowledgments

The authors would like to acknowledge the assistance of Dr. Kaye Lathrop from Los Alamos Laboratories who provided the computer program and computational facilities needed to complete the shielding analyses.

References

1. S. J. RIMSHAW and E. E. KETCHEN, "Curium Data Sheets," ORNL-4357, Oak Ridge National Laboratory (1969).
2. T. ROCKWELL, Ed., Reactor Shielding Design Manual, United States Atomic Energy Commission, McGraw-Hill Book Co., Inc. (1956).
3. E. D. ARNOLD, "Handbook of Shielding Requirements and Radiation Characteristics of Isotope Power Sources for Terrestrial Marine and Space Applications," ORNL-3576, Oak Ridge National Laboratory, (1964).

Footnotes

- a. This code, TWOTRAN, was developed by Kaye Lathrop of Los Alamos Laboratories.

Table I
Properties of Cm_2O_3

Mass Density	9.0 gm/cm ³
Power Density	2.65 watts/gm
Melting Point	2200°C
Boiling Point	3700°C

Table II
Radiological Properties of ^{244}Cm

α Particle Energy	~	5.8 MeV
α Particle Rate		$1.1(10^{12})$ particles/watt second
γ Particle Energy	~	0.4 MeV
γ Particle Rate		$3.267(10^8)$ γ /watt second
Neutron Energy	~	1.5 MeV
Neutron Rate		$4.07(10^6)$ n/watt second

Table III

NEUTRON CALCULATIONS

Calc. No.	Separation Distance ft.	LiH Thickness in.	U^{238} Thickness in.	Reps with Cd 1/16 in.	Reps without Cd 0 in.
1	3	3	1/4	$0.536(10^5)$	$0.998(10^5)$
2	3	6	1/4	$0.363(10^5)$	$0.733(10^5)$
3	3	9	1/4	$0.194(10^5)$	$0.380(10^5)$
4	2	6	1/4	$0.318(10^5)$	$0.688(10^5)$
5	2	6 front out to water 1 wrapped around source	1/4	$0.0773(10^5)$	$0.2623(10^5)$
6	2	6 water moved closer into source radially	1/4	$0.9696(10^5)$	$0.2546(10^5)$

Source Strength = 10^{18} neutrons \leq 35 kw for 2000 hrs.

Table IV

γ CALCULATIONS

Calc. No.	Separation Distance ft.	LiH Thickness in.	U^{238} Thickness in.	Dose Rads
7	2	3	1/4	$3.71(10^5)$
8	3	3	0	$4.55(10^5)$
9	3	3	1/2	$6.67(10^4)$
10	3	3	1/4	$1.64(10^5)$

Source Strength = $8.21(10^8)_{\gamma} \leftrightarrow 35$ kw for 2000 hrs.

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7. Representative Geometry for Curium Shield Calculations
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9. Curium Shield Characteristics

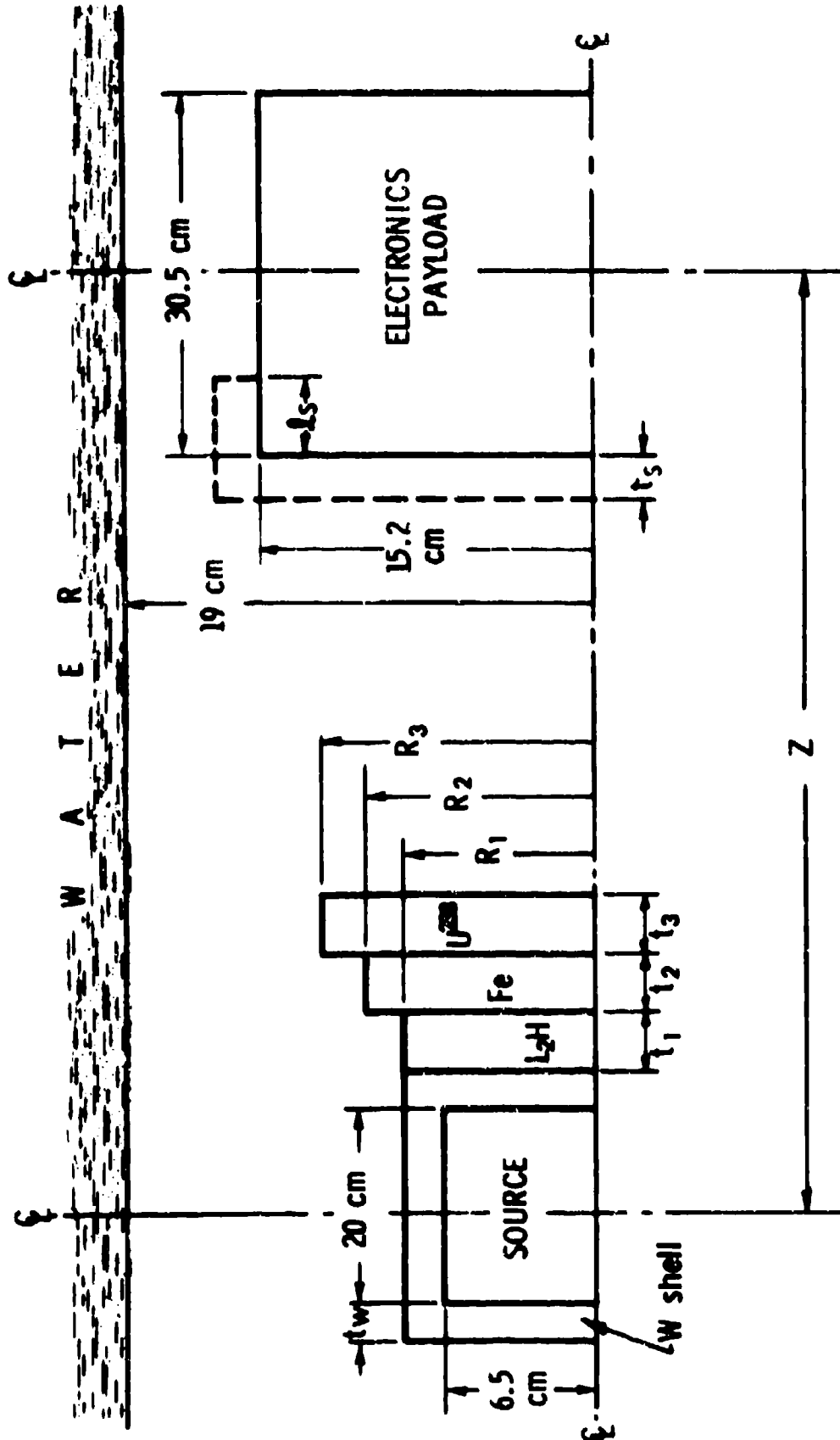


Fig. 1

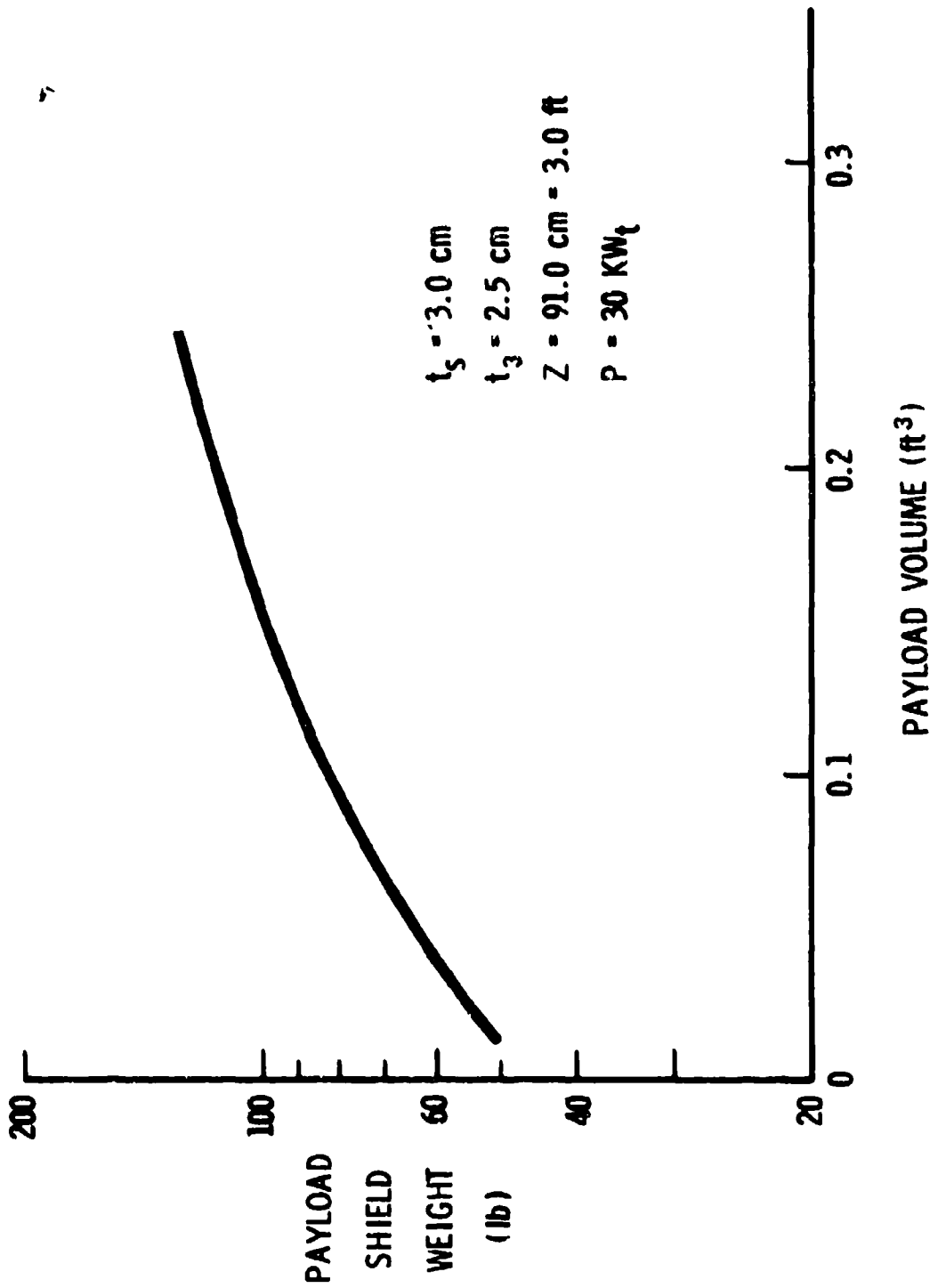


Fig. 2

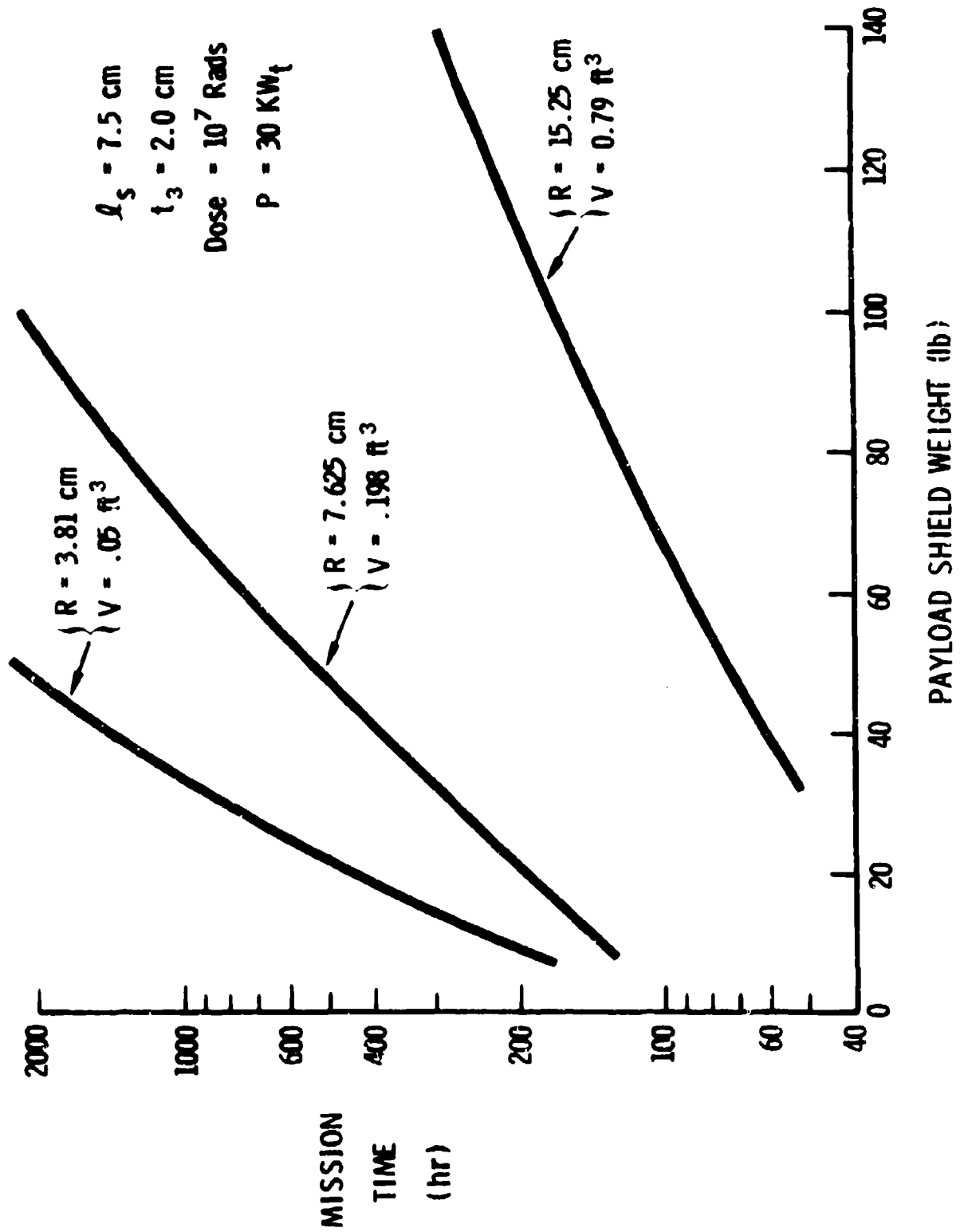


Fig. 3

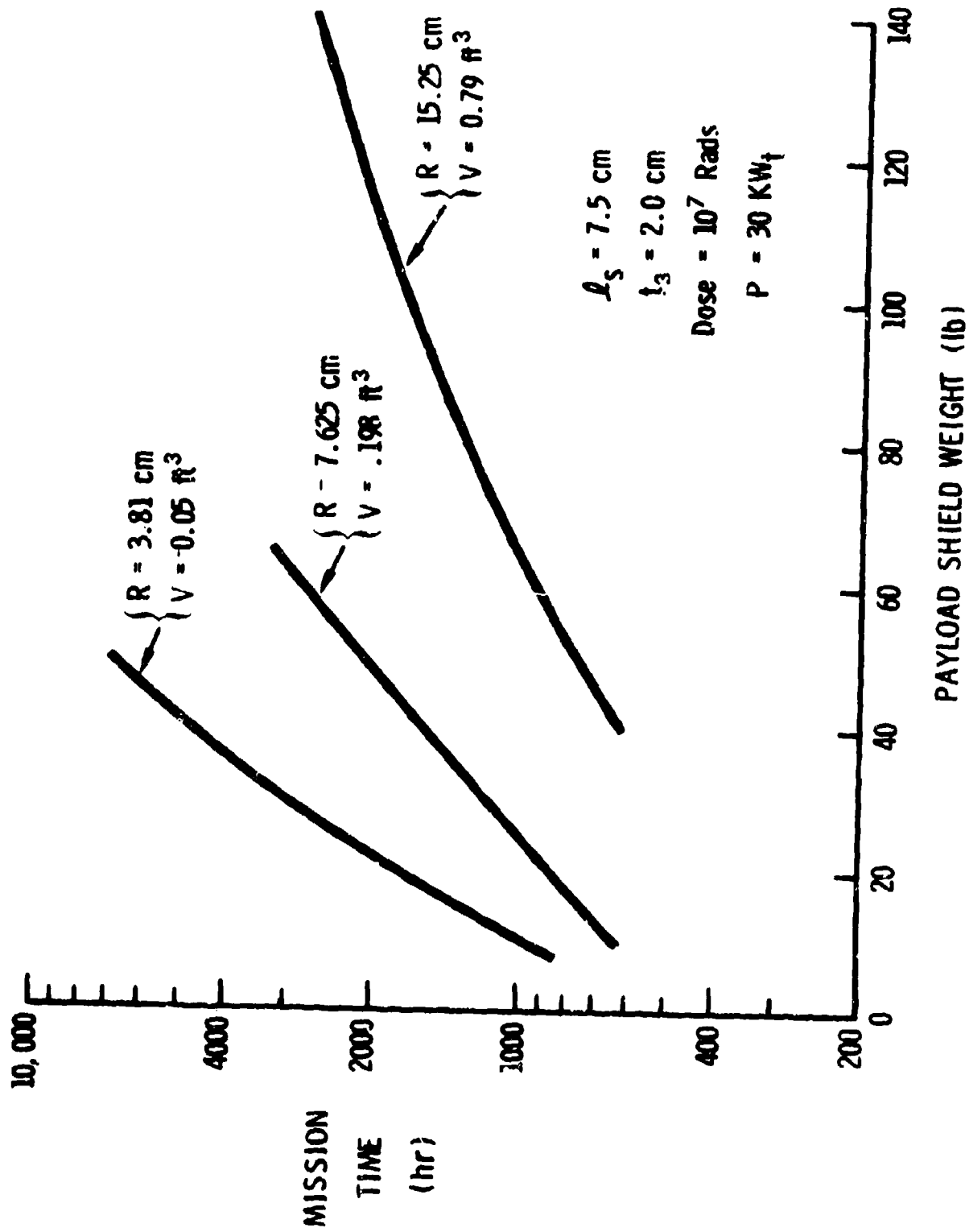


Fig. 4

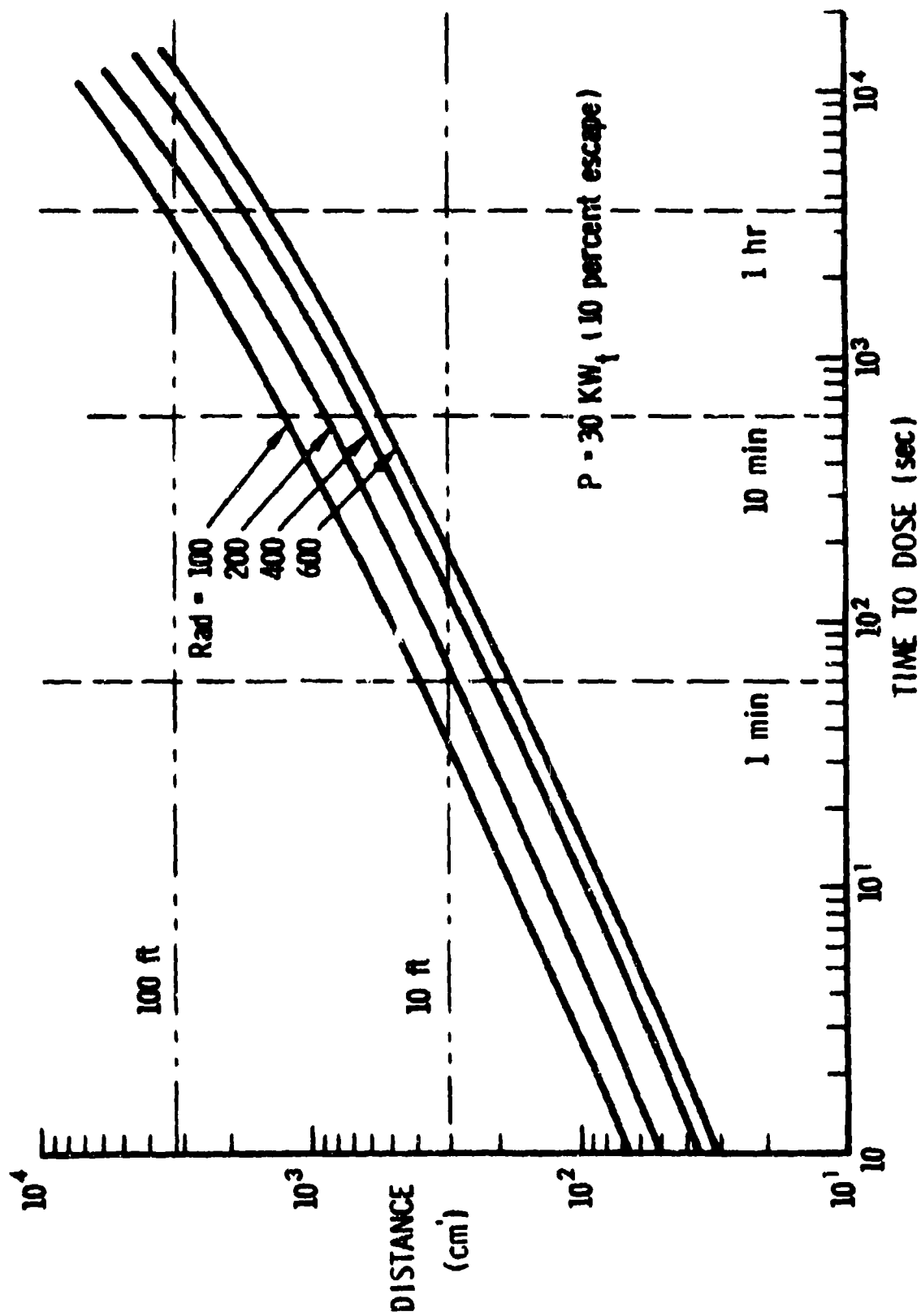


Fig. 5

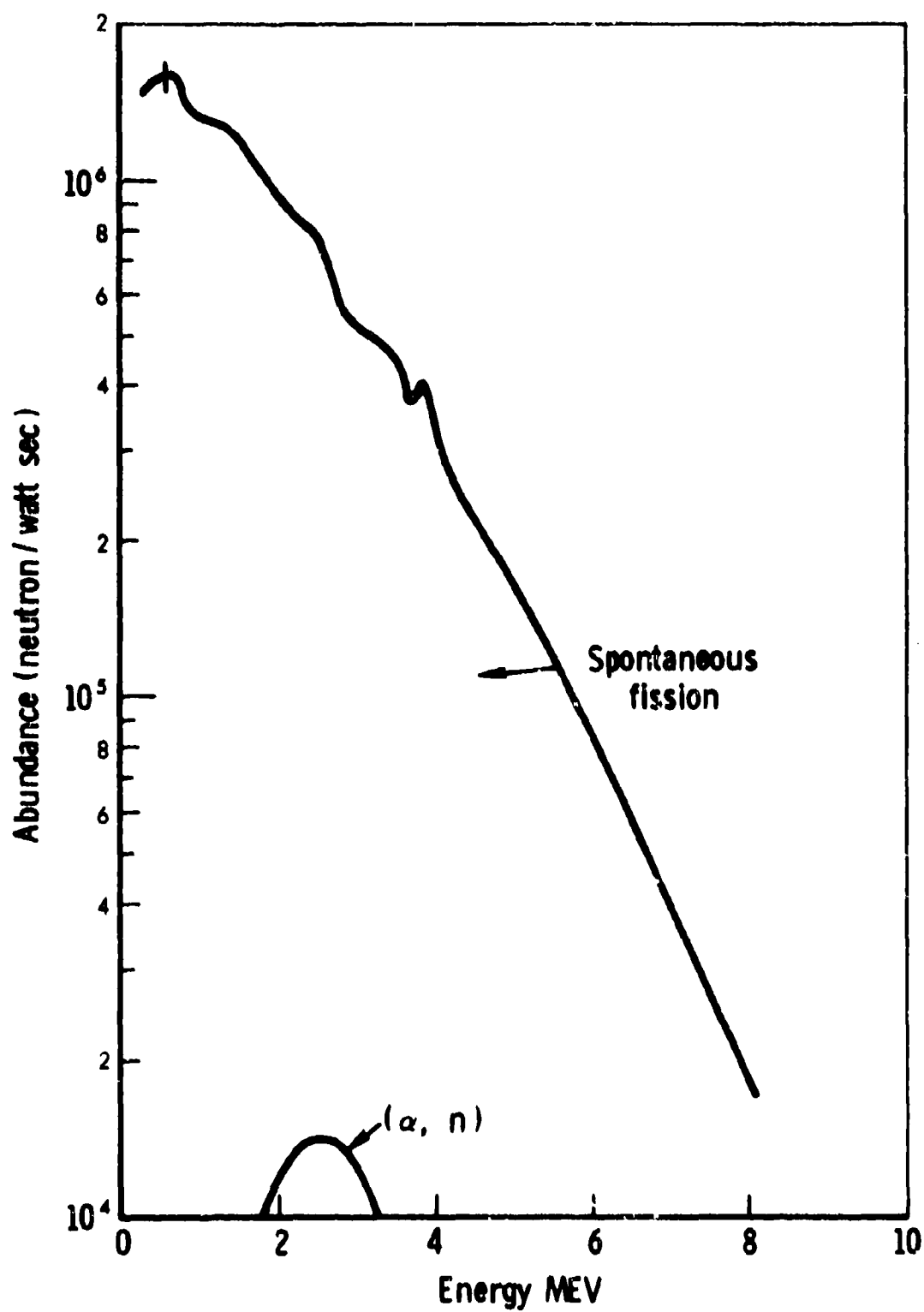


Fig. 6

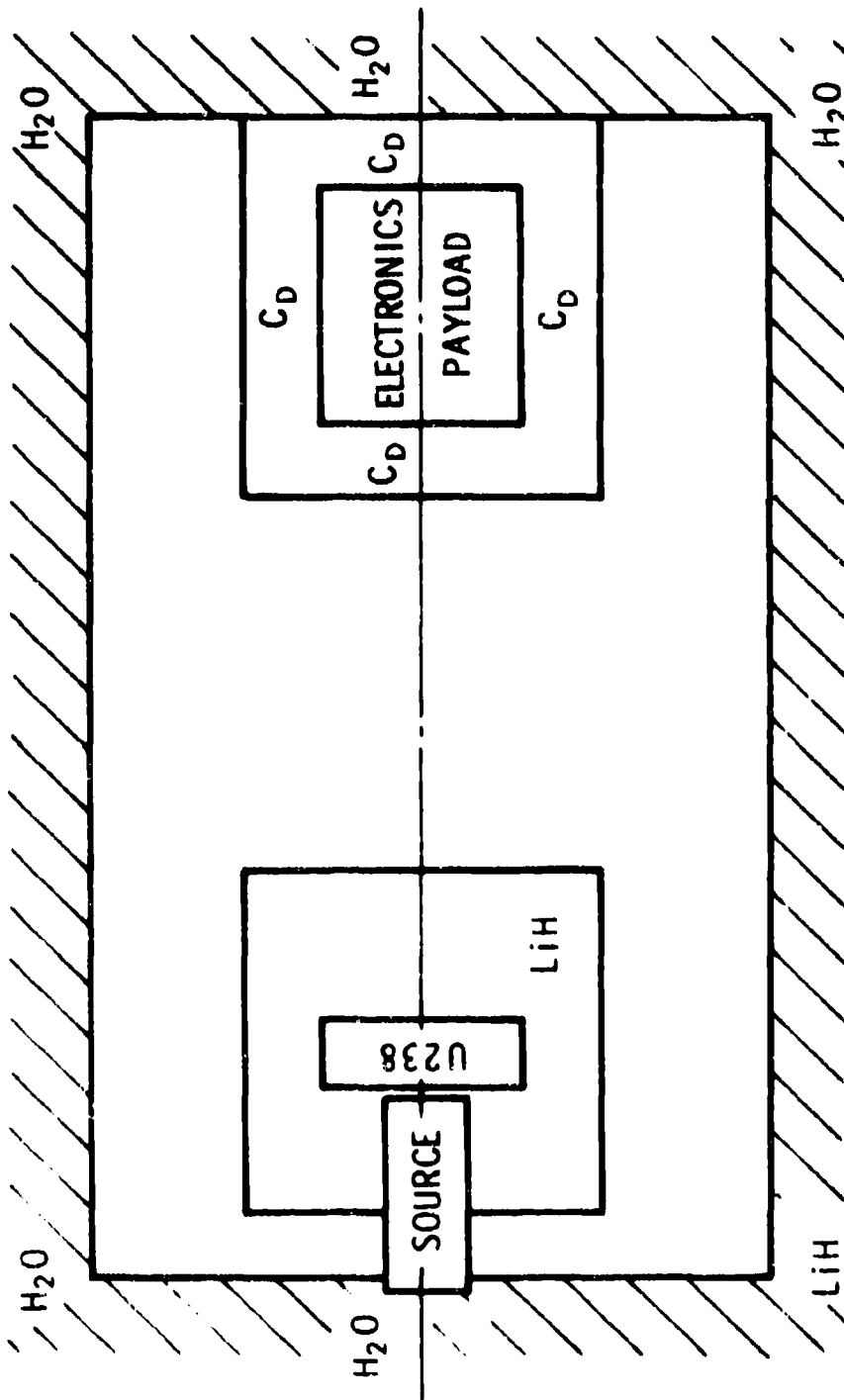


Fig. 7

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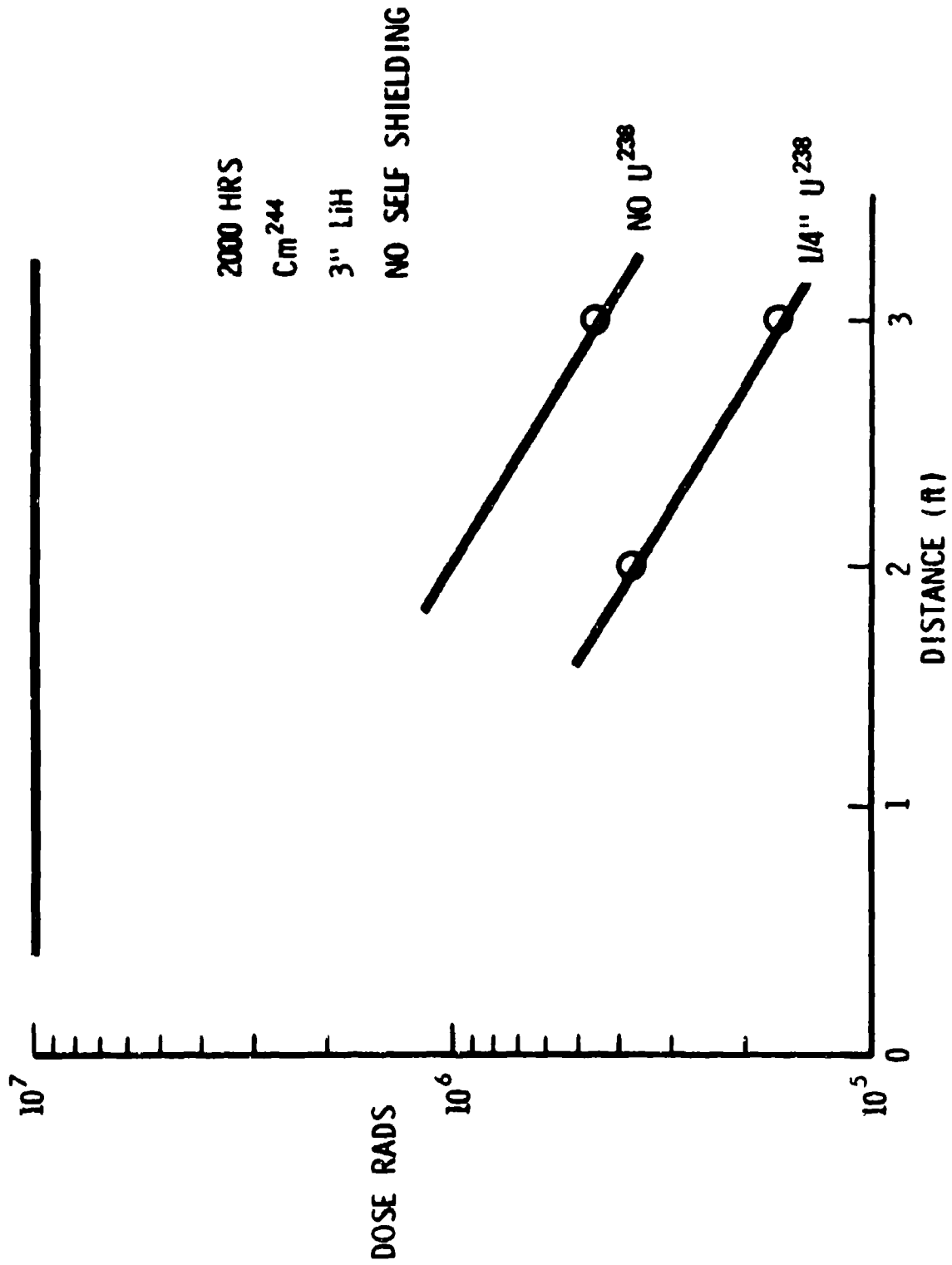


Fig. 8

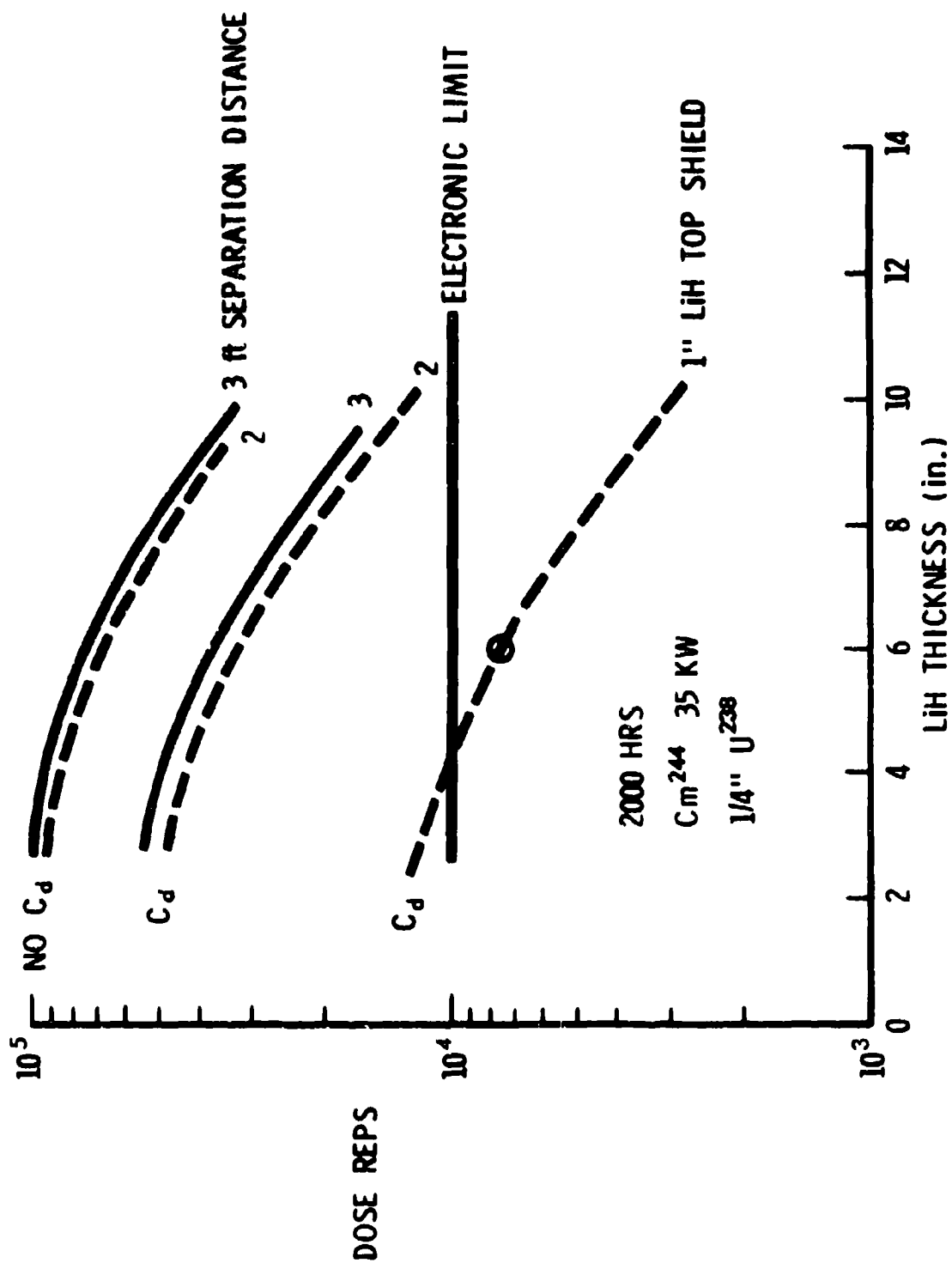


Fig. 9